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Technical note

# Kinetic parameters of a low enriched uranium fuelled material test research reactor at end-of-life

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A B S T R A C T

The kinetic parameters at end-of-life of a material test reactor fuelled with low enriched uranium fuel were calculated. The reactor used for the study was the IAEA's 10 MW benchmark reactor. Simulations were carried out to calculate core excess reactivity, neutron flux spectrum, prompt neutron generation time and effective delayed neutron fraction. Nuclear reactor analysis codes including WIMS-D4 and CITATION were employed to carry out these calculations. It was observed that in comparison with the beginning-of-life values, at end-of-life, the neutron flux increased throughout the core, the prompt neutron generation time increased by 3.68% while the effective delayed neutron fraction decreased by 0.35%.



## 1. Introduction

A large number of research reactors around the world, which were fuelled with HEU based fuels having uranium enrichment around 90% in $^{235}$U isotope have been converted to use LEU based fuels having uranium enrichment of 20% in $^{235}$U isotope, since 20% fuel enrichment is an isotopic barrier for weapon usability (Glaser, 2005). The IAEA has devised a standard benchmark MTR (IAEA-TECDOC-233, 1980) in order to facilitate reactor conversion. Many theoretical calculations have been performed and reported in various documents (IAEA-TECDOC-233, 1980; IAEA-TECDOC-643, 1992).

All the calculations reported in the IAEA's guidebooks (IAEA-TECDOC-233, 1980; IAEA-TECDOC-643, 1992) deal with the reactor behaviour at the beginning of reactor core life. However, the reactor remains at the beginning-of-life (BOL) just for a while and as soon as the fission reaction starts, the isotopic concentration in the fuel changes. Since most of the neutronic parameters depend on the fuel composition, these parameters also change with the change in the fuel isotopic concentration. Hence, need is always felt to find different reactor parameters at the end-of-life (EOL).

The work presented in this paper deals with the calculation of the neutron flux, prompt neutron generation time and effective delayed neutron fraction the 10 MW IAEA benchmark reactor (IAEA-TECDOC-643, 1992) using LEU fuel at the EOL.

## 2. Reactor description

The reactor analyzed is the same one utilized for the benchmark problem solved in IAEA-TECDOC-233, with the water in the central flux trap replaced with a 7.7 cm × 8.1 cm block of aluminum containing a square hole 5.0 cm on each side (IAEA-TECDOC-643, 1992). Description of the low enriched uranium core of the reactor is given in Table 1. The core configuration and burn up of fuel elements in percentage of loss of the number of initial $^{235}$U atoms at BOL is given in Fig. 1 while that of EOL is given in Fig. 2. Other details can be found in the reference documents (IAEA-TECDOC-233, 1980; IAEA-TECDOC-643, 1992).

## 3. Analysis procedure

### 3.1. Reactor simulation codes

The WIMS-D4 (Halsall, 1980) code was used for the generation of group constants for various core regions while CITATION (Fowler et al., 1971) was used to perform global core calculations. Detailed description of these codes can be found in related material.

### 3.2. Simulation methodology

The CITATION code was used to calculate various core parameters like $k_{eff}$, neutron fluxes and adjoint fluxes. The core was simulated in the $x$–$y$–$z$ geometry. All control rods were assumed

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**Table 1**
Data for low enriched uranium core of IAEA 10 MW benchmark reactor (IAEA-TECDOC-643, 1992).

| Parameter | Value |
|---|---|
| Active core height | 60.0 cm |
| Extrapolation length | 8.0 cm (in 8.0 cm distance from the core, the cosine-shaped flux goes to zero) |
| Space at the grid plate per fuel element | 7.7 cm × 8.1 cm |
| Fuel element cross-section | 7.6 cm × 8.05 cm including support plate |
| Meat dimensions | 6.3 cm × 0.051 cm × 60.0 cm |
| Thickness of support plate | 0.475 cm |
| Number of fuel plates per fuel element | 23 identical plates, each 0.127 cm thick |
| Number of fuel plates per control element | 17 identical plates, each 0.127 cm thick |
| Identification of the remaining plate positions of the control element | Four plates of aluminum, each 0.127 cm thick in the position of the first, the third, the twenty-first, and the twenty-third standard plate position; water gaps between the two sets of aluminum plates |
| Specifications of the LEU ($UAl_x$-Al) fuel | a.Enrichment 90 w/o U-235 in HEU, 20 w/o U-235 in LEU<br>b.390 g U-235 ($\rho_U$ = 4.40 g/cm$^3$) per fuel element (23 plates)<br>c.72 w/o of uranium in the $UAl_x$-Al<br>d.Only U-235 and U-238 in the fresh fuel |
| Total power | 10 $MW_{th}$ |
| Xenon-State | Homogeneous Xenon content corresponding to average-power-density |
| Nominal coolant flow rate (m$^3$/h) | 1000 |
| Coolant inlet temperature (°C) | 38 |
| Hot channel factors | a.Radial × local power peaking factor = 1.4<br>b.Axial power peaking factor = 1.5<br>c.Engineering factor = 1.2<br>d.Overpower factor = 1.2 |

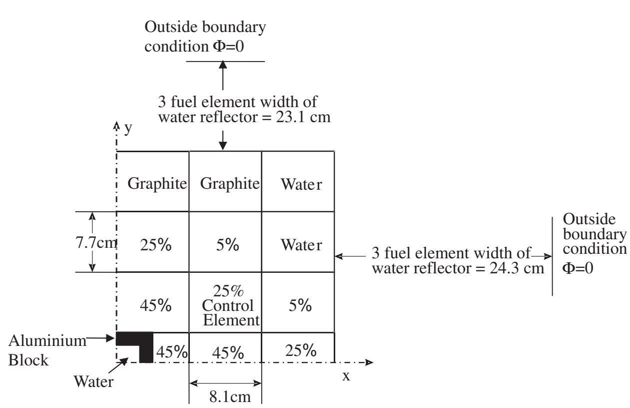


**Fig. 1.** IAEA 10 MW benchmark reactor configuration at BOL as defined in IAEA-TECDOC-233, 1980 and IAEA-TECDOC-643, 1992.

to be fully withdrawn. The fuelled and non-fuelled portions of each standard and control fuel element were modeled separately. The WIMS-D4 code was used for computation of macroscopic absorption cross-section ($\Sigma_a$), the $\nu$-fission cross-section ($\nu\Sigma_f$), the diffusion coefficient *(D)*, the scattering matrix ($\Sigma_{s,g\rightarrow g'}$) and the fission spectrum for all groups. These data are required by CITATION as input. Five energy groups were used to perform multi-group calculations. These groups are given in Table 2. The simulation methodology has been fully described and validated in our recent works (Muhammad and Majid, 2008, 2009).

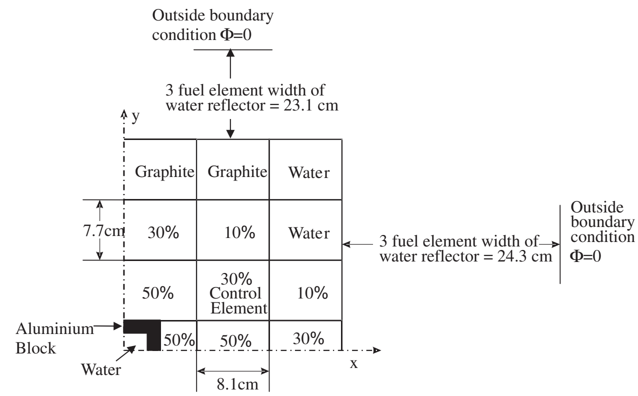


**Fig. 2.** IAEA 10 MW benchmark reactor configuration at EOL as defined in IAEA-TECDOC-233, 1980 and IAEA-TECDOC-643, 1992.

**Table 2**
Energy groups used for macroscopic cross-section generation by WIMS-D4.

| Group no. | $E_U$ (eV) | $E_L$ (eV) | Group type[a] | Flux type[b] |
|---|---|---|---|---|
| 1 | $10.0 \times 10^6$ | $0.821 \times 10^6$ | Fast | Fast |
| 2 | $0.821 \times 10^6$ | $5.530 \times 10^3$ | Resolved resonances | |
| 3 | $5.530 \times 10^3$ | 0.625 | Unresolved resonances | Epithermal |
| 4 | 0.625 | 0.14 | Thermal | Thermal |
| 5 | 0.14 | 0 | Thermal | |

[a] Muhammad and Majid, 2008.
[b] IAEA-TECDOC-233, 1980.

## 4. Results and discussions

### 4.1. The $k_{eff}$ at EOL

The CITATION code gives the value of $k_{eff}$ directly. This value for the given core configuration at EOL was found to be 0.999560 showing that the reactor fuel had actually come to the end of its life (Table 3).

### 4.2. Neutron flux spectrum

The shape of neutron fluxes of all the groups is same (Fig. 3) at BOL and EOL but the magnitude of flux of all the three groups increases at EOL (Fig. 4). The thermal neutron flux at the mid of the central flux trap also increases (Table 4) to $2.2316 \times 10^{14}$/cm$^2$ s from $2.1780 \times 10^{14}$/cm$^2$ s showing an increase of 2.48% from the BOL value calculated using same method and technique (Muhammad and Majid, 2008). The reason behind this increase is that the quantity of fuel decreases with burn up decreasing the fuel

**Table 3**
Comparison of different parameters at BOL and EOL.

| Parameter | BOL | | | | | EOL[f] |
|---|---|---|---|---|---|---|
| | Ref. 1[a] | Ref. 2[b] | Ref. 3[c] | Ref. 4[d] | Ref. 5[e] | |
| $k_{eff}$ | 1.01823 | – | – | 1.02462 | 1.018273 | 0.999560 |
| $\Lambda$ (μs) | 43.74 | – | 44.39 | 45.42 | 44.03 | 45.65 |
| $\beta_{eff}$ | 0.007275 | 0.00732 | 0.007219 | 0.007351 | 0.007185 | 0.007160 |

[a] IAEA-TECDOC-643, 1992 (Appendix G-1).
[b] IAEA-TECDOC-643, 1992 (Appendix G-2).
[c] IAEA-TECDOC-643, 1992 (Appendix G-3).
[d] IAEA-TECDOC-643, 1992 (Appendix G-4).
[e] Muhammad and Majid, 2008.
[f] This work.

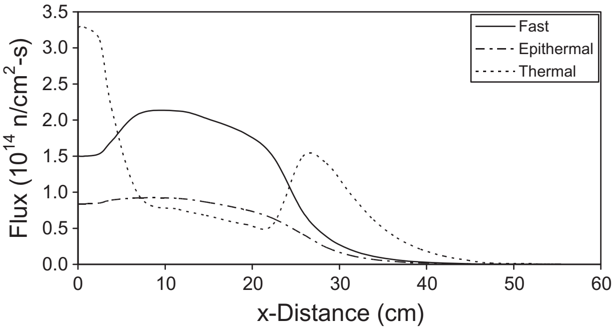


**Fig. 3.** Three-group neutron flux in the reactor.

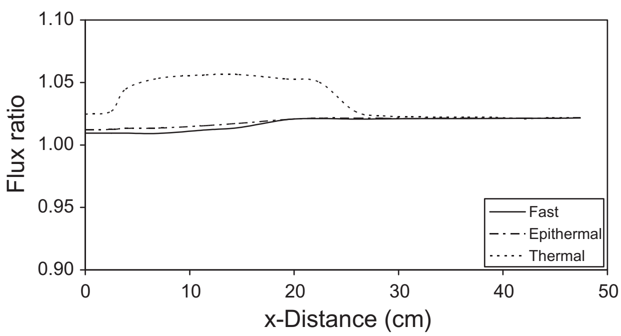


**Fig. 4.** Ratio of neutron fluxes at EOL to that at BOL.

**Table 4**
Comparison of neutron flux at mid of central flux trap at BOL and EOL ($10^{14}$ n/cm$^2$ s).

| Flux type | BOL[a] | EOL[b] | % increase from BOL to EOL |
|---|---|---|---|
| Fast | 1.789 | 1.806 | 0.95 |
| Epithermal | 0.963 | 0.975 | 1.25 |
| Thermal | 2.178 | 2.232 | 2.48 |

[a] Muhammad and Majid, 2008.
[b] This work.

number density and since the power remains constant at 10 MW, the neutron flux increases.

### 4.3. The prompt neutron generation time $\Lambda$

The prompt neutron generation time ($\Lambda$) was calculated from the following relation (Ott and Neuhold, 1985).

$$\Lambda = \overline{\left(\frac{1}{\mathrm{v}}\right)}\frac{1}{\nu\sum_f} \tag{1}$$

where $\overline{\left(\frac{1}{\mathrm{v}}\right)}$ is the average inverse neutron velocity given by (Ott and Neuhold, 1985):

$$\overline{\left(\frac{1}{\mathrm{v}}\right)} = \frac{\int_v \int_E \frac{1}{\mathrm{v}(E)}\varphi(r,E)dEdV}{\int_\mathrm{v} \int_E \varphi(r,E)dEdV} = \frac{\sum_{g=1}^{5}\left(\frac{1}{\mathrm{V}_g}\right)\varphi_g}{\sum_{g=1}^{5}\varphi_g} \tag{2}$$

where $\varphi_g$ is the group flux and $\mathrm{V}_g$ is the group average neutron velocity, and

$$\nu\sum_f = \frac{\int_V \int_E \nu\sum_f(r,E)\,\varphi(r,E)\,dE\,dV}{\int_V \int_E \varphi(r,E)\,dE\,dV} = \frac{\sum_{g=1}^{5}\nu\sum_{fg}\varphi_g}{\sum_{g=1}^{5}\varphi_g} \tag{3}$$

the required data were obtained from WIMS and CITATION.

The value of $\Lambda$ at EOL was found to be 45.65 µs (Table 3) showing an increase of 3.68% over its value at BOL (Muhammad and Majid, 2008).

During the reactor operation, the total amount of fissile isotopes ($^{235}$U and $^{239}$Pu) decreases linearly with burnup (Fig. 5) (The concentration of $^{241}$Pu is too low to have any significant effect). This linear decrease is seen in the value of $\nu\sum_f$ as shown in Fig. 6 (especially in groups 4 and 5 where most of the fission takes place), which also decreases linearly with fuel burnup. The increased value of $\Lambda$ reflects the decreased value of $\nu\sum_f$ and increased thermalization of the neutron flux at EOL.

### 4.4. The effective delayed neutron fraction $\beta_{eff}$

The $\beta_{eff}$ was calculated from the following relation (IAEA-TEC-DOC-643, 1992).

$$\beta_{eff} = \frac{\sum_{g=1}^{5}\varphi_g^{+}\chi'_g}{\sum_{g=1}^{5}\varphi_g^{+}\chi_g}\cdot\frac{\sum_i\beta^i\sum_{g=1}^{5}(\nu\Sigma_f)_g^i\varphi_g^I}{(\nu\Sigma_f\varphi)^T} \tag{4}$$

where $i$ is an index for fissionable isotopes, $\chi'_g$ is $g$-group fraction in typical delayed neutron spectrum, $\chi_g$ is $g$-group fraction of fission spectrum, $\beta^i$ is $i$-nuclide delayed neutron fraction, $\varphi_g^{+}$ is core-average adjoint flux for group g,$\varphi_g^I$ is core-integrated $g$-group ordinary flux, $(\nu\sum_f\varphi)^T$ is total source of fission neutrons. The values of $\chi'_g$ and $\beta^i$ were taken from reference (Keepin, 1965.) while the remaining required data were obtained from WIMS and CITATION. The value of $\beta_{eff}$ at EOL was found to be 0.007160 (Table 3) showing a decrease of 0.35% over its value at BOL (Muhammad and Majid, 2008). The amount of different isotopes changes from BOL to EOL (Fig. 5) resulting in change in the value of $\beta_{eff}$. Since the amount of Pu isotopes and their change in concentration with burnup is small as compared to the amount of $^{235}$U isotope, the change in the value of $\beta_{eff}$ is also small.

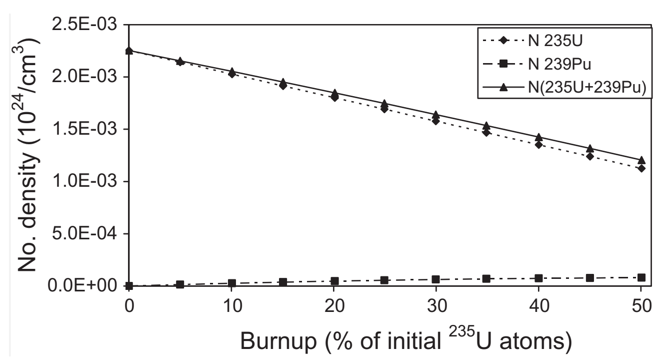


**Fig. 5.** No. densities of different isotopes with fuel burnup.

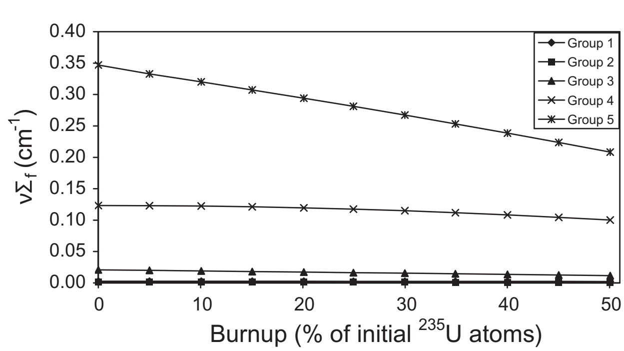


**Fig. 6.** Variation of $\nu\sum_f$ with fuel burnup.

## 5. Conclusions

The kinetic parameters of an MTR fuelled with LEU fuel change with fuel burnup from the BOL to EOL. The changed parameters govern the worth of control rods and the importance of any reactivity insertion or removal. The transient behaviour and the inherent safety features of the reactor are also dependent on the calculated kinetic parameters. Hence, the reactor will have different behaviour at EOL than that at BOL. Also, the neutron flux throughout the reactor increases at EOL which will reduce the time of isotope production.

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